

NON-PUBLIC?: N
ACCESSION #: 9307190055
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 1 PAGE: 1 OF 07

DOCKET NUMBER: 05000220

TITLE: Automatic Reactor Scram on High Neutron Flux Signal
Received During Surveillance Testing due to Personnel
Error
EVENT DATE: 01/26/93 LER #: 93-002-01 REPORT DATE: 07/07/93

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Mr. Robert Tessier, Manager TELEPHONE: (315) 349-2707
Maintenance NMP1

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On January 26, 1993 at 0936 hours, Nine Mile Point Unit 1 (NMP1) received an automatic reactor scram initiation signal. Specifically, while performing a monthly calibration test on Average Power Range Monitor (APRM) Flow Converters, a high neutron flux trip signal was inadvertently initiated on Reactor Protection System (RPS) Channel 12 while a manual halfscram signal was inserted on RPS Channel 11. At the time of the event, the plant was operating at 100 percent rated thermal power with reactor pressure at 1030 pounds per square inch gauge (psig) and reactor temperature at 531 degrees Fahrenheit.

The root cause for this event has been determined to be personnel error by failure to follow a procedure.

Immediate operator actions included commencing scram recovery activities and initiating a controlled plant cooldown. Additional corrective actions included: 1) improving management expectations during pre-job briefing activities; 2) revising supervisory roles during surveillance testing and; 3) issuing a Lessons Learned Transmittal.

END OF ABSTRACT

TEXT PAGE 2 OF 7

I. DESCRIPTION OF EVENT

On January 26, 1993 at 0936 hours, Nine Mile Point Unit 1 (NMP1) received an automatic reactor scram initiation signal. At the time of the event, the plant was operating at 100 percent rated thermal power with reactor pressure at 1030 pounds per square inch gauge (psig) and reactor temperature at 531 degrees Fahrenheit.

Immediately prior to the scram, Instrument and Controls (I&C) technicians were performing a monthly calibration test on Average Power Range Monitor (APRM) flow converters R103A and R103B (Reactor Protection System Channels 11 and 12, respectively), located on Main Control Room G Panel. The converters receive input signals from flow transmitters located in each Reactor Recirculation loop downstream of the Recirculation pump discharge, and are designed to monitor Reactor Recirculation flow rates.

Preliminary action step 7.1.13 (calibration of Channel 11 Flow Converter R103A) to Instrument Surveillance Procedure N1-ISP-032-004, "Reactor Recirculation Flow Converter Calibration," directs the I&C technician to notify the Chief Shift Operator (CSO) to manually initiate a half-scram signal on RPS Channel 11. Upon completion of the manual half-scram insertion, the calibration procedure provides technicians with two alternate courses of action; one for Reactor Recirculation core flow less than 20 percent, and the other for Reactor Recirculation core flow at greater than 20 percent.

During this test sequence, RPS Channel 12 Converter R103B mode switch was switched from the "Operate" to the "Test" position. When the flow signal was removed from the RPS Channel 12 APRMs, a high neutron flux signal was immediately generated by these APRMs. This caused an automatic half-scram in RPS Channel 12 and, concurrent with the manual half-scram in RPS Channel 11, brought in a full scram. Control Room annunciators F4-1-8, "RPS Ch 12 Reactor Neutron Monitor," and F4-2-8, "RPS Ch 12 Auto Reactor Trip," on F panel confirmed that a neutron monitoring trip signal had been received on RPS Channel 12.

Plant parameters at the time of the event were as follows: all five Reactor Recirculation System pumps (Nos. 11, 12, 13, 14, and 15) were running providing approximately 95 percent of rated core flow; Feedwater System alignment was utilizing motor-driven Feedwater pump No. 11 and shaft-driven Feedwater pump No. 13 (Feedwater pump No. 12 in standby).

Following the scram signal, all control rods were confirmed to have inserted to position 00. The immediate plant response was a rapid decrease in reactor vessel water inventory, resulting in actuation of an automatic half-scram signal in RPS Channel 11 and a High Pressure Coolant Injection (HPCI) System initiation signal, which were expected plant responses. Control Room operators entered Emergency Operating Procedure N1-EOP-2, "RPV Control," to re-establish vessel water inventory and Special Operating Procedures N1-SOP-1, "Reactor Scram," to assist in establishing plant stability and system recovery.

TEXT PAGE 3 OF 7

I. DESCRIPTION OF EVENT (cont.)

Lowest reactor vessel water level recorded during the transient was plus 21 inches (which is 105 inches above the top of active fuel).

The Main Turbine tripped approximately 10 seconds after the scram (which was 5 seconds after completion of a full automatic reactor scram signal), with the generator trip occurring 5 seconds after that. The Turbine bypass valves opened momentarily to control reactor pressure, then reclosed. Feedwater System pump No. 12 was started in anticipation of a low reactor vessel water level and a HPCI actuation.

Reactor vessel water level recovery resulted in water level reaching 93 inches. Feedwater System pump No. 12 tripped at this point, which was not an expected evolution. Feedwater pump No. 12 has tripped two previous times during scram recoveries.

The reactor scram was reset at 0939 hours. Normal reactor vessel water level was reestablished at approximately 0945 hours. HPCI was reset at 0956 hours. All other reactor parameters exhibited a normal response during the transient.

II. CAUSE OF EVENT

A root cause investigation was performed utilizing Nuclear Interfacing Procedure NIP-ECA-02, "Root Cause Evaluation." The root cause of this event was determined to be personnel error by failure to follow a procedure.

The I&C technicians who were performing Instrument Surveillance Procedure N1-ISP-032-004, incorrectly performed procedural steps 7.2.1 through 7.2.5. Specifically, in preparing to record RPS Channel 11 Flow Converter R103A "As Found" values, the procedure directs the test individual(s) to record Reactor Recirculation flow. Additionally, if the Recirculation flow rate is less than 20 percent, the procedure directs test personnel to take RPS Channel 12 Flow Converter R103B mode switch to the "Test" position. If flow rate is greater than 20 percent, procedure steps involving RPS Channel 12 should be entered as "Not Applicable" (N/A) and these steps in the procedure should not be performed.

With the plant's Reactor Recirculation flow rate at approximately 95 percent and with a manual half-scam signal inserted in RPS Channel 11, technicians placed RPS Channel 12's Flow Converter mode switch in the "Test" position, initiating a full scram signal.

Contributing causes to this event were:

Verbal Communication - The pre-job briefing by I&C supervision with the I&C technicians did not address potential problems which may be encountered during the test. The pre-job brief also failed to address any past problems which may have emerged during previous performances of N1-ISP-032-004.

TEXT PAGE 4 OF 7

II. CAUSE OF EVENT (cont.)

Work Practices - Prior to performing the undesired procedural steps, the I&C technicians failed to use the self-checking process to the level necessary to prevent this event. Both technicians questioned each other concerning the performance of steps on RPS Channel 12, however, I&C supervision was never consulted to resolve their concerns.

The trip of Feedwater System pump No. 12 upon reactor vessel high water level during post-scam recovery was unanticipated. An evaluation has concluded that the pump trip was the result of downstream Feedwater System low flow control bypass valve for pump No. 12 not indicating fully closed at the time of the high water level signal. The cause for this valve open indication was an out of adjustment valve shut limit switch resulting in the shut limit switch contact points not making up.

III. ANALYSIS OF EVENT

This event is reportable in accordance with 10CFR 50.73 (a)(2)(iv), which

requires the Licensee to report "any event or condition that resulted in the manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)."

An automatic reactor scram initiation on a high neutron flux signal is a designed Reactor Protection System (RPS) function.

The RPS consists of two independent logic channels (Channels 11 and 12). The output of the two logic channels are combined so that they both must be tripped to initiate a scram. A high neutron flux trip limits the heat flux to a level well below that which could cause fuel damage. The integrity of the fuel clad as a barrier to the release of fission product is assured when fuel clad safety limits are not exceeded. In this event, actual neutron flux level did not increase. Each channel trip was a result of a signal generated by a manual initiation of the trip logic for both RPS Channels 11 and 12.

The High Pressure Coolant Injection (HPCI) System is an operating mode of the Feedwater System. The initiation of the HPCI System on low reactor water level is a design function to provide adequate cooling to the reactor core.

The reactor scram on a high neutron flux signal was a conservative plant response. There were no significant safety concerns as a result of this event, nor was the reactor in an unsafe condition. The reactor scram posed no safety consequences to the health and safety of the general public or plant personnel.

An evaluation was completed on the tripping of Feedwater System pump No. 12 during the scram recovery evolution. This concluded that pump No. 12 fulfilled its HPCI System function during the event by starting on manual initiation (would also have auto started on a

TEXT PAGE 5 OF 7

III. ANALYSIS OF EVENT (cont.)

valid initiation signal) at the beginning of the transient and with Feedwater System pump No. 11, helped to restore reactor vessel water level. Feedwater pump No. 12 tripped on a high reactor vessel water level signal and a low flow control valve open indication to prevent vessel overfill. In the event water level had again dropped to 53 inches, pump No. 12 would have restarted.

The turbine trip protective circuitry worked as designed. The turbine trip occurred 5 seconds after completion of the automatic scram logic.

It took an additional 5 seconds because a manual scram signal was inserted in RPS channel 11 when the full scram occurred.

The Final Safety Analysis Report (FSAR) was reviewed to determine the consequences of the additional time delay (10 seconds versus 5 seconds) for the turbine trip. Two FSAR analyses; 1) Chapter VII, "Engineered Safeguards," which addresses HPCI performance during a small line break in the drywell; and 2) Chapter XV, "Safety Analysis on a Loss of Feedwater Transient," were reviewed for effects of this 5 second delay on the assumptions. In both cases, it was concluded that an additional 5 second delay would not affect the outcome of these analyses in a non-conservative manner.

IV. CORRECTIVE ACTIONS

The immediate corrective actions were to perform all scram recovery actions, place the plant in a stable condition, and determine the cause of the scram. Also, a Deviation/Event Report (DER No. 1-93-0203) was written to evaluate the event and provide disposition.

Additional corrective actions to assess overall plant impact and prevent recurrence include:

1. Near Term Corrective Actions

A. The individual controlling the test evolution was removed from the list of qualified performers of Instrument Surveillance Procedure N1-ISP-032-004. Remediation Training and Evaluation will be conducted prior to reinstatement.

B. Improved pre-job briefing guidelines have been prepared and presented to Maintenance Department supervisors. Management expectations regarding pre-job briefing and self/peer verification fundamentals have been presented to the Maintenance Departments.

C. Supervisory coverage will be provided for all I&C procedures that involve halfscram initiations, ESF actuations, or have the potential for significant impact on generation capacity. This coverage will be required until long term corrective actions A and B are completed.

TEXT PAGE 6 OF 7

IV. CORRECTIVE ACTIONS (cont.)

D. A Lessons Learned Transmittal documenting the significance, consequences, and corrective actions from this event has been prepared and communicated to Maintenance Department personnel.

E. The time delay adjustment on the turbine protective circuit relay was tested and found to be set at 5 seconds.

F. A Work Order (W.O. #1182583-00) has been issued to disassemble and rebuild the Feedwater System low flow control bypass valve. This work is scheduled for completion during NMP1's present Refuel Outage.

2. Long Term Corrective Actions

A. A revision of the current training program, focusing on upgrading technical knowledge in high risk surveillance/Preventive Maintenance (PM) activities, is scheduled for completion by March 31, 1994.

B. A revision of the On-The-Job-Training (OJT)/On-The-Job-Evaluation ((OJE) process with a focus on individual qualification signoffs and technical knowledge requirements. This action is scheduled to be completed by March 31, 1994.

C. A continuing evaluation (via work place self-assessment) will be performed to verify that the concepts presented in the Maintenance Department performance principles are being adhered to. This evaluation is scheduled for completion by December 31, 1993.

V. ADDITIONAL INFORMATION

A. Failed components: none.

B. Previous similar events:

Nine Mile Point Unit 1 has experienced previous events related to personnel errors (reference LER 90-26). A review of corrective actions taken as a result of these events has concluded that some actions have been less than adequate in reducing personnel error occurrences. These conclusions have been identified to plant management (reference Deviation Event Report No. C-93-0170).

To further review and evaluate the process used in the development

of corrective actions, a Quality Assurance Department surveillance audit was conducted to assess the adequacy of the root cause evaluation process (reference Deviation Event Report

TEXT PAGE 7 OF 7

V. ADDITIONAL INFORMATION (cont.)

No. 1-93-0161). This also identified weak areas in this process which may have contributed to the development of previous event ineffective corrective actions. These Deviation Event Reports will be used to document and implement improvements/enhancements to the site's corrective action process.

C. Identification of components referred to in this LER:

IEEE 803 EHS IEEE 805
COMPONENT FUNCTION SYSTEM ID

Reactor Protection System N/A JC
Feedwater System N/A SJ
High Pressure Coolant Injection System N/A BJ
Reactor Recirculation System N/A BJ
Reactor Pressure Vessel RPV SB
Pump P AD, SJ
Neutron Monitor MON IG
Flow Converter CNV AD
Average Power Range Monitor MON IG
Main Turbine TRB TA
Generator GEN TB
Flow Transmitters FT AD
Annunciators (Control Room) ANN IB

*** END OF DOCUMENT ***
